ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT

UNIT NUMBER 1, CYCLE 9

STARTUP TEST REPORT

PREPARED BY PLANT REACTOR ENGINEERING GROUP

APPROVED:

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DISK: CYCLES/8

IE26
111

TABLE OF CONTENTS

1.0 INTRODUCTION

The Joseph M. Farley Unit 1 Cycle 9 Startup Test Report addresses the core refueling and the startup tests required by plant procedures following the refueling. The report provides a brief synopsis of each test and gives a comparison of measured parameters with design predictions, Technical Specifications, or values assumed in the FSAR.

Unit 1 of the Joseph M. Farley Nuclear Plant is a Westinghouse three loop pressurized water reactor rated at 2652 MWth. The core loading consists of 157 17 x 17 fuel assemblies. The Unit began commercial operations on December 1, 1977.

Cycle 8 power operation began on December 2, 1986, and ended on March 25, 1988, with an average core burnup of 16,190 MWD/MTU.

2.0 UNIT 1, CYCLE 9 CORE REFUELING

REFERENCES

1. Westinghouse Refueling Procedure FP-ALA-R8

2. Westinghouse WCAP 11755 (The Nuclear Design and Core Management of the Joseph M. Farley Unit 1 Power Plant Cycle 9)

The fuel shuffle commenced on 4/3/88 and was completed in 24 days on 4/27/88. The as-loaded Cycle 9 core is shown in Figures 2.1 through 2.4, which give the location of each fuel assembly and insert, including the burnable poison insert locations and configurations. The burnable poison inserts used for Cycle 9 are wet annular burnable absorber rods (WABAs). The Cycle 9 core has a nominal design lifetime of 17500 MWD/MTU and consists of 5 region 6 assemblies, 22 region 9A assemblies, 4 region 9B assemblies, 58 region 10 assemblies, 40 region 11A assemblies, and 28 region 11B assemblies. Fuel assembly inserts include 48 full length control rod clusters, 52 wet annular burnable poison inserts, two secondary sources, and 55 thimble plug inserts.

During core unload, a detailed fuel inspection program was conducted to eliminate leaking and defective fuel assemblies. Each fuel assembly was visually inspected with binoculars and was ultrasonically leak tested with the Brown Bovari Failed Fuel Rod Detection System (FFRDS). Assemblies having observed or suspected visual defects were re-examined with an underwater TV camera, and assemblies considered suspect on the initial leak test were subjected to re-examination with the FFRDS System. Five assemblies were found to have leaking fuel rods, of which J18 and K44 had been scheduled for core reload. Nine assemblies had visual defects, of which J18, K44, J05, J17, K19, K21, and J25 were scheduled for core reload. These assemblies were rejected from core reload, which required a revised Cycle 9 core design to be obtained from Westinghouse.

FIGURE 2.1:UNIT 1 CYCLE 9 REFERENCE LOADING PATERN

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							300 J21	290 J24	32 J16						
2					66 J29	K33 2A03	4W160 2A43	R27 K09	4W140 2A50	R15 2A15	720 J59				
3				230 J02	37 2A18	16W290 2A28	R37 K37	16W540 2A52	R21 K38	16W310 2A29	38 2A02	88 J49			
4			18 J34	[illegible]47 F44	12W580 2A61	R19 K50	16W250 2A37	SS05 K48	16W440 2A08	R23 K61	12W570 2A62	R29 K22	50 J23		
5		40 J55	170 2A23	12W540 2A42	R38 F35	16W320 2A24	140 K33	710 K25	13 K12	16W430 2A20	R03 F46	12W530 2A67	610 2A21	65 [illegible]	
6		R17 2A14	16W420 2A17	R10 K27	16W490 2A01	R32 K13	16W470 2A45	R01 K45	16W300 2A56	R35 K40	16W270 2A27	R39 K01	16W530 2A13	[illegible]	
7	241 J39	6W150 2A57	R13 K47	16W410 2A34	390 K53	16W580 2A53	R48 K28	560 K57	R11 K39	16W350 2A54	370 K73	16W590 2A37	R36 K11	4W110 2A49	360 J20
8	28 K56	R04 K03	16W500 2A51	410 K06	100 K46	R24 K31	99 K26	540 K36	650 K51	R12 K49	510 K14	47 K52	16W600 2A41	R20 K04	690 K18
9	660 J44	4W130 2A60	R09 K60	16W370 2A46	200 K24	16W570 2A48	R26 K10	450 K05	R02 K58	16W550 2A47	24 K42	16W260 2A4[illegible]	330 K07	4W120 2A65	410 J33
10		R44 2A25	16W480 2A22	R46 K32	16W540 2A31	R31 K30	16W330 2A64	R43 K02	16W510 2A59	R45 K59	16W380 2A09	R18 K16	16W400 2A36	R14 2A07	
11		36 J46	8 2A16	12W600 2A68	R34 F43	16W390 2A30	620 K54	110 K55	380 K08	16W340 2A19	R07 F50	12W550 2A58	190 2A33	75 J65	
12			3 J08	R05 K[illegible]	12W560 2A55	R41 K20	16W520 2A35	SS06 K29	16W460 2A39	R25 K15	12W590 2A63	R22 K34	150 J13		
13				250 J48	56 2A04	16W280 2A38	R06 K17	16W450 2A46	R28 K41	16W360 2A05	31 2A11	4 J28			
14					74 J64	R08 2A06	4W090 2A66	R40 K35	4W100 2A44	R42 2A12	46 J50				
15							45 J01	60 J07	12 J15						

<-- FUEL ASSEMBLY INSERT SERIAL NUMBER

<-- FUEL ASSEMBLY SERIAL NUMBER

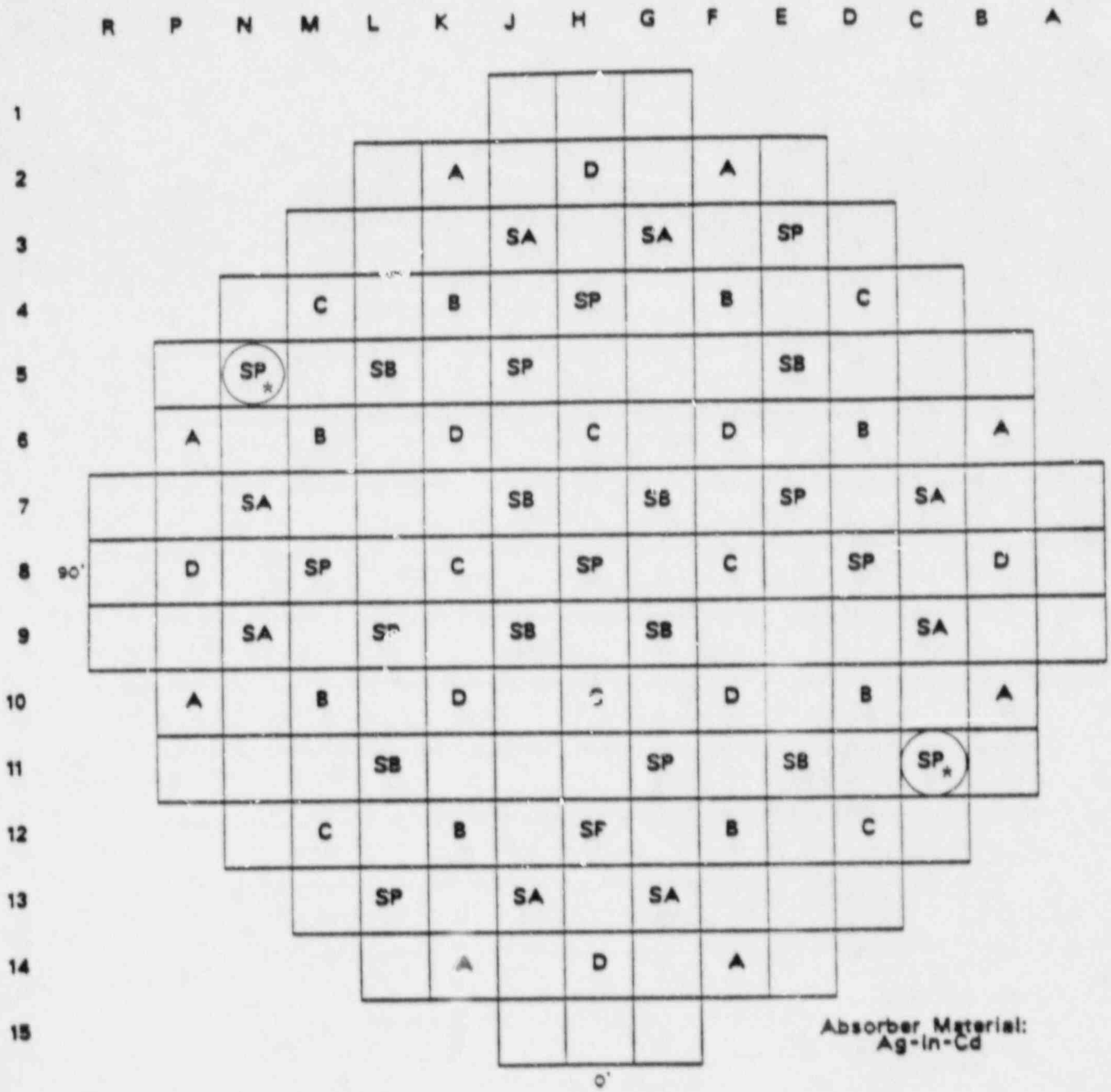
NORTH

The original w/o U-235 enrichments were:

Region 6 (F) assemblies 2.995%
Region 9A (J) assemblies 3.597%
Region 9B (J) assemblies 3.906%
Region 10 (K) assemblies 3.597%
Region 11A (2A) assemblies ... 3.805%
Region 11B (2A) assemblies ... 4.207%

FIGURE 2.2

CONTROL ROD LOCATIONS



FUNCTION	NUMBER OF CLUSTERS
Control Bank D	8
Control Bank C	8
Control Bank B	8
Control Bank A	8
Shutdown Bank SB	8
Shutdown Bank SA	8
SP (Spare Rod Locations)	13

* LOCATION N5 & C11 = CORE WATER LEVEL THERMOCOUPLE PROBES

FIGURE 2.3

BURNABLE ABSORBER AND SOURCE ASSEMBLY LOCATIONS

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1															
2							4		4						
3						16		16		16					
4					12		16	SS	16		12				
5				12		16				16		12			
6			16		16		16		16		16		16		
7		4		16		16				16		16		4	
8 90°			16										16		
9		4		16		16				16		16		4	
10			16		16		16		16		16		16		
11				12		16				16		12			
12					12		16	SS	16		12				
13						16		16		16					
14							4		4						
15															

0°

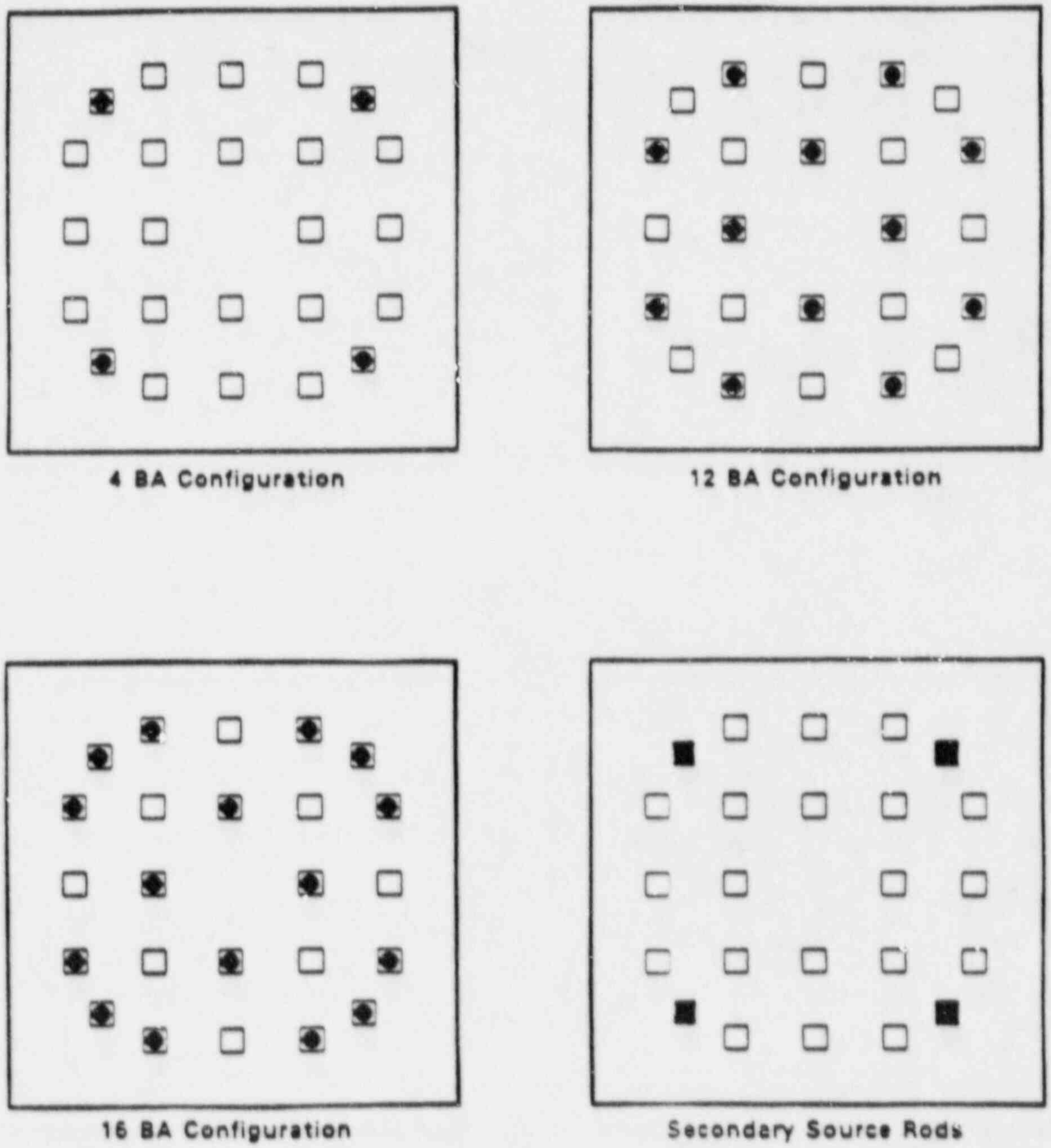
##	Number of WABAs
SS	Secondary Source

704 WABAs in
52 Clusters

(WABA = WET ANNULAR BURNABLE ABSORBER)

FIGURE 2.4

BURNABLE ABSORBER AND SECONDARY SOURCE ROD CONFIGURATIONS



4 BA Configuration

12 BA Configuration

16 BA Configuration

Secondary Source Rods

3.0 CONTROL ROD DROP TIME MEASUREMENT (FNP-1-STP-112)

PURPOSE

The purpose of this test was to measure the drop time of all full length control rods under hot full-flow conditions in the reactor coolant system to ensure compliance with Technical Specification requirements.

SUMMARY OF RESULTS

For the hot full-flow condition ($T_{avg} \geq 541°$ and all reactor coolant pumps operating) Technical Specification 3.1.3.4 requires that the rod drop time from the fully withdrawn position shall be < 2.2 seconds from the beginning of stationary gripper coil voltage decay until dashpot entry. All full length rod drop times were measured to be less than 2.2 seconds. The longest drop time recorded was 1.85 seconds for rod B-6. The rod drop time results for both dashpot entry and dashpot bottom are presented in Figure 3.1. Mean drop times are summarized below:

TEST CONDITIONS	MEAN TIME TO DASHPOT ENTRY	MEAN TIME TO DASHPOT BOTTOM
Hot full-flow	1.63 sec	2.20 sec

To confirm normal rod mechanism operation prior to conducting the rod drops, a Control Rod Drive Test (FNP-0-ETP-3643) was performed. In the test, the stepping waveforms of the stationary, lift and movable gripper coils were examined, and the functioning of the digital rod position indicator and the bank overlap unit were checked. Rod stepping speed measurements were also conducted. All results were satisfactory.

NORTH

UNIT 1 CYCLE 9

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R								90°							
P						1.64 2.14		1.60 2.12		1.63 2.14					
N							1.64 2.17		1.62 2.14						
M				1.60 2.12		1.63 2.12				1.63 2.14		1.63 2.19			
L					1.60 2.17						1.62 2.18				
K		1.73 2.27		1.59 2.15		1.62 2.14		1.57 2.12		1.63 2.15		1.65 2.23		1.65 2.19	
J			1.61 2.13				1.56 2.05		1.55 2.07				1.61 2.13		
H	0°	1.62 2.20				1.60 2.12				1.60 2.10				1.60 2.08	180°
G			1.68 2.17				1.55 2.06		1.56 2.05				1.64 2.15		
F		1.72 2.22		1.67 2.22		1.62 2.13		1.57 2.08		1.58 2.10		1.64 2.17		1.60 2.13	
E					1.60 2.18						1.61 2.18				
D				1.64 2.21		1.61 2.13				1.56 2.08		1.62 2.17			
C							1.64 2.18		1.66 2.17						
B						1.67 2.23		1.70 2.21		1.85 2.3[illegible]					
A								270°							

DRIVE LINE "DROP TIME" TABULATION

TEMPERATURE - 549°F PRESSURE - 2235 psig % FLOW - 100

X.XX	BREAKER "OPENING" TO DASHPOT ENTRY - IN SECONDS
X.XX	BREAKER "OPENING" TO DASHPOT BOTTOM - IN SECONDS

DATE - 5-18-88

4.0 INITIAL CRITICALITY (FNP-1-ETP-3601)

PURPOSE

The purpose of this procedure was to achieve initial reactor criticality under carefully controlled conditions, establish the upper flux limit for the conduct of zero power physics tests, and operationally verify the calibration of the reactivity computer.

SUMMARY OF RESULTS

Initial reactor criticality for Cycle 9 was achieved during dilution mixing at 1855 on May 18, 1988. The reactor was allowed to stabilize at the following critical conditions:

RCS pressure	2244 psig
RCS temperature	547°F
Intermediate range power	1.0×10^{-8} amp
RCS boron concentration	1977 ppm
Control Bank D position	182 steps

Following stabilization, the point of adding nuclear heat was determined and a checkout of the reactivity computer using both positive and negative flux periods was successfully accomplished. In addition, source and intermediate range neutron channel overlap data were taken during the flux increase preceding and immediately following initial criticality to demonstrate that adequate overlap existed.

5.0 ALL-RODS-OUT ISOTHERMAL TEMPERATURE COEFFICIENT AND BORON ENDPOINT MEASUREMENT (FNP-1-ETP-3601)

PURPOSE

The objectives of these measurements were to determine the hot, zero power isothermal and moderator temperature coefficients for the all-rods-out (ARO) configuration and to measure the ARO boron end-point concentration.

SUMMARY OF RESULTS

The measured ARO, hot zero power temperature coefficients and the ARO boron endpoint concentration are shown in Table 5.1. The isothermal temperature coefficient was measured to be -0.32 pcm/°F which meets the design acceptance criteria. This gives a calculated moderator temperature coefficient (corrected to ARO) of +1.954 pcm/°F which is within the Technical Specification limit of +5.0 pcm/°F. Thus, no rod withdrawal limits were needed to satisfy the +5.0 pcm/°F limit. The design acceptance criterion for the ARO critical boron concentration was also satisfactorily met.

TABLE 5.1

ARO, HZP ISOTHERMAL AND MODERATOR TEMPERATURE COEFFICIENT

Rod Configuration	Boron Concentration	Measured α_T	Calculated α_{mod}	α_T Design Acceptance Criterion
	ppm	pcm/°F	pcm/°F	pcm/°F
Bank D at 209 Steps	1991	-0.32	+1.954	-0.52 ± 2

α_T - Isothermal temperature coefficient, includes -2.19 pcm/°F doppler coefficient

α_{mod} - Moderator only temperature coefficient, corrected to the all-rods-out condition

ARO, HZP BORON

ARO BORON ENDPOINT CONCENTRATION

Rod Configuration	Measured C_B (ppm)	Design - predicted C_B (ppm)
Bank D at 228 Steps	1996.4	1998 ± 50

6.0 CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS (FNP-1-ETP-3601)

PURPOSE

The objective of the bank worth measurements was to determine the integral reactivity worth of each control and shutdown bank for comparison with the values predicted by design.

SUMMARY OF RESULTS

The rod worth measurements were performed using the bank interchange method in which: (1) the worth of the bank having the highest design worth (designated as the "Reference Bank") is carefully measured using the standard dilution method; and (2) the worths of the remaining control and shutdown banks are derived from the change in reference bank reactivity needed to offset full insertion of the bank being measured.

The control and shutdown bank worth measurement results are given in Table 6.1. The measured worths satisfied the review criteria both for the banks measured individually and for the combined worth of all banks.

TABLE 6.1

SUMMARY OF CONTROL AND SHUTDOWN BANK WORTH MEASUREMENTS

Bank	Predicted Bank Worth & Review Criteria (pcm)	Measured Bank Worth (pcm)	Percent Difference
Control A	559 ± 100	591.8	5.9
Control B (Ref.)	1227 ± 123	1182.4*	-3.6
Control C	787 ± 118	739.3	-6.1
Control D	1049 ± 157	1001.7	-4.5
Shutdown A	1039 ± 156	1028.2	-1.0
Shutdown B	919 ± 138	849.1	-7.6
All Banks Combined	5580 ± 558	5392.5	-3.36

*Measured by dilution method

7.0 POWER ASCENSION PROCEDURE (FNP-1-ETP-3605)

PURPOSE

The purpose of this procedure was to provide controlling instructions for:

1. NIS intermediate and power range setpoint changes, as required prior to startup and during power ascension.
2. Ramp rate limitation during power ascension.
3. Conduct of startup and power ascension testing, to include:
   a. HZP reactor physics tests (FNP-1-ETP-3601).
   b. incore movable detector system alignment (FNP-1-ETP-3606).
   c. incore/excore AFD channel recalibration (FNP-1-STP-121).
   d. core hot channel factor surveillance (FNP-1-STP-110).
   e. reactor coolant system flow measurement (FNP-1-STP-115.1).

SUMMARY OF RESULTS

In order to satisfy Technical Specification requirements for invoking special core physics test exceptions, preliminary trip setpoints of less than or equal to 25% power were used for the NIS intermediate and power range channels. When physics tests were completed, the power range setpoint was increased to 80% to enable power escalation (above 25%) for calorimetric recalibration of the power range channels. (The 80% setpoint was used instead of 109% in case the uncalibrated power range channels were indicating nonconservatively.) At approximately 30% power, the power range channels were recalibrated, the high-range trip setpoint was restored to 109%, and setpoint currents were determined for the intermediate range channels.

The Westinghouse fuel warranty limits the power ramp rate to 3% of full power per hour between 20% and 100% power until full power has been sustained for 72 cumulative hours out of any seven-day operating period. This ramp rate was observed during the ascension to 100% power.

The startup and power ascension test program controlled by ETP-3605 commenced with the zero-power physics tests described in Sections 4.0 - 6.0 of this report. Following physics testing, incore movable detector system core limit settings were determined for all modes of operation during the ascension to 30% power. The Incore-Excore recalibration test (described in Section 8.0) was performed at 30%, 48% and 80% power, and the reactor coolant system flow measurement (Section 9.0) was performed at 100% power. Surveillance of reactor core hot channel factors was accomplished using data from the full-core flux maps taken during the Incore-Excore procedure. As summarized in Table 7.1, all results were within Technical Specification limits.

TABLE 7.1

SUMMARY OF POWER ASCENSION FLUX MAP DATA

Parameters	Map 209	Map 215	Map 216
Date	05/23/88	05/25/88	05/27/88
Time	17:40	02:33	03:12
Avg. % Power	31.86	48.02	80.16
Max FΔH	1.5680	1.5563	1.5020
Max. Power Tilt*	1.0262	1.0262	1.0176
Avg. Core % A. O.	+6.825	+9.244	+4.652
Maximum FQ(Z)	2.1543	2.1499	1.9581
FQ Limit	4.6052	4.5240	2.8783
Xenon Conditions	Non-Equilibrium	Equilibrium	Equilibrium

*Calculated (incore) power tilts based on assembly FΔHN from all assemblies.

8.0 INCORE-EXCORE DETECTOR CALIBRATION (FNP-1-STP-121)

PURPOSE

The objective of this procedure was to determine the relationship between power range upper and lower excore detector currents and incore axial offset for the purpose of calibrating the delta flux penalty to the overtemperature ΔT reactor trip, and for calibrating the control board and plant computer axial flux difference (AFD) channels.

SUMMARY OF RESULTS

A full Incore-Excore recalibration, consisting of one full-core base-case flux map and five quarter-core flux maps, was performed at approximately 30% power. The six maps were obtained at axial offsets of +6.8%, +23%, +12.4%, -5.4%, -12.0%, and -26.0% and, during each map, detector currents and calorimetric data were taken. The detector currents were normalized to 100% power and a least-squares fit was performed to derive an output current vs. axial offset equation for each top and bottom detector. Calibration currents derived from these equations were used to recalibrate the nuclear instrumentation system (NIS) delta flux channels and the overtemperature ΔT delta flux input.

Power was then increased to 48% and stabilized in preparation for a full core flux map for hot channel factor surveillance. However, excore quadrant power tilt ratic (QPTR) calculations performed during the power ascension disclosed that a high indicated QPTR had developed. Since the flux map indicated that core hot channel factors were satisfactory and that the high excore QPTR was not associated with the actual incore tilt (which was in a different quadrant), a detector equation I-zero current renormalization was performed to correct the problem and the NIS delta flux instrumentation was recalibrated.

When power escalation was resumed, the QPTR again begin to increase, reaching 1.02 at just above 99% power. Therefore, following a planned power reduction and stabilization at 80% for maintenance purposes, a full-core flux map was performed. Again, the flux map indicated that core hot channel factors were satisfactory and that the indicated high QPTR was not a reflection of actual incore tilt. Therefore, the detector equation I-zero current renormalization was repeated and the NIS delta flux instrumentation was recalibrated. No further calibration problems were experienced during a subsequent further power reduction for turbine DEH checkout, nor during the return to full power.

The finalized incore-excore equations used to derive delta flux channel calibration data are given in Table 8.1.

TABLE 8.1

DETECTOR CURRENT VERSUS AXIAL OFFSET EQUATIONS
OBTAINED FROM INCORE-EXCORE CALIBRATION TEST

CHANNEL N41:

I-Top	=	0.6902*AO	+	178.92 μa
I-Bottom	=	-1.2255*AO	+	175.36 μa

CHANNEL 42:

I-Top	=	0.7932*AO	+	173.46 μa
I-Bottom	=	-1.2903*AO	+	174.42 μa

CHANNEL N43:

I-Top	=	0.7254*AO	+	180.01 μa
I-Bottom	=	-1.3113*AO	+	189.48 μa

CHANNEL N44:

I-Top	=	0.7355*AO	+	167.49 μa
I-Bottom	=	-1.3021*AO	+	170.73 μa

9.0 REACTOR COOLANT SYSTEM FLOW MEASUREMENT (FNP-1-STP-115.1)

PURPOSE

The purpose of this procedure was to measure the flow rate in each reactor coolant loop in order to confirm that the total core flow met the minimum flow requirement given in the Unit 1 Technical Specifications.

SUMMARY OF RESULTS

To comply with the Unit 1 Technical Specifications, the total reactor coolant system flow rate measured at normal operating temperature and pressure must equal or exceed 265,500 gpm for three loop operation. From the average of ten calorimetric heat balance measurements, the total core flow was determined to be 282,826 gpm, which meets the above criterion.

NT-88-0430

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Alabama Power
the southern electric system

September 19, 1988

Docket No. 50-348

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Joseph M. Farley Nuclear Plant - Unit 1
Cycle 9 - Startup Report

Gentlemen:

Enclosed is the Startup Report for Unit 1 Cycle 9 as outlined in the April 19, 1988 letter from Mr. R. P. McDonald.

If you have any questions, please advise.

Yours very truly,

W. G. Hairston, III

W. G. Hairston, III

WGH/MDR:emb

Enclosure

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IE26
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